B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. RCS component joints are made by welding, bolting, rolling, or pressure loading. Valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

AEC GDC Criterion 16 (Ref. 1), requires means for monitoring the reactor coolant pressure boundary to detect LEAKAGE. LCO 3.4.16, "RCS Leakage Detection Instrumentation," describes requirements for leakage detection instrumentation.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

BASES (continued)

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The USAR (Ref. 2) analysis for SGTR assumes the plant has been operating with a 5 gpm primary to secondary leak rate for a period of time sufficient to establish radionuclide equilibrium in the secondary loop. Following the tube rupture, the initial primary to secondary LEAKAGE is relatively inconsequential when compared to the mass transfer through the ruptured tube.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 1 gpm (at 70°F) primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the reactor coolant pressure boundary (RCPB). LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

Seal welds are provided at the threaded joints of all reactor vessel head penetrations (spare penetrations, full-length Control Rod Drive Mechanisms, and thermocouple columns). Although these seals are part of the RCPB as defined in 10CFR50 Section 50.2, minor leakage past the seal weld is not a fault in the RCPB or a structural integrity concern. Pressure retaining components are differentiated from leakage barriers in the ASME Boiler and Pressure Vessel Code. In all cases, the joint strength is provided by the threads of the closure joint.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere

BASES

LCO

c. Identified LEAKAGE (continued)

with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified leakage must be evaluated to assure that continued operation is safe. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One Steam Generator (SG)

The 150 gallons per day (gpd) limit on one SG is based on implementation of the Steam Generator Voltage Based Alternate Repair Criteria and is more restrictive than standard operating leakage limits to provide additional margin to accommodate a crack which might grow at greater than the expected rate or unexpectedly extend outside the thickness of the tube support plate.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.15, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

BASES (continued)

ACTIONS

A.1

Unidentified LEAKAGE in excess of the LCO limits must be identified or reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1, B.2.1, and B.2.2

If unidentified LEAKAGE cannot be identified or cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals, gaskets, and pressurizer safety valves seats is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours. If the LEAKAGE source cannot be identified within 54 hours, then the reactor must be placed in MODE 5 within 84 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

C.1, C.2.1, and C.2.2

If RCS identified LEAKAGE, other than pressure boundary leakage, is not within limits, then the reactor must be placed in MODE 3 within 6 hours. In this condition, 14 hours are allowed to reduce the identified leakage to within limits. If the identified LEAKAGE is not

BASES

ACTIONS

C.1, C.2.1, and C.2.2 (continued)

within limits within this time, the reactor must be placed in MODE 5 within 44 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions in an orderly manner without challenging plant systems.

D.1 and D.2

If RCS pressure boundary LEAKAGE exists or if SG LEAKAGE (150 gpd limit) is not within limits, the reactor must be placed in MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

The RCS water inventory balance must be met with the reactor at steady state operating condition (stable temperature, power level, equilibrium xenon, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). Therefore, a Note is added allowing that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by monitoring containment atmosphere radioactivity. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.16, "RCS Leakage Detection Instrumentation."

The 24 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.14.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.14.2 (continued) Generator Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.
REFERENCES	1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criterion 16, issued for comment July 10, 1967, as referenced in USAR, Section 1.2. 2. USAR, Section 14.5.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

RCS PIVs separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.14, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.14.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressurization of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 1) that identified potential intersystem LOCAs as a significant

BASES

BACKGROUND (continued)	contributor to the risk of core melt. A subsequent study (Ref. 2) evaluated various PIV configurations to determine the probability of intersystem LOCAs. PIVs are provided to isolate the RCS from low pressure systems susceptible to intersystem LOCAs. The PIVs are listed in the LCO section of these Bases. Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.
APPLICABLE SAFETY ANALYSES	Reference 1 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the Residual Heat Removal (RHR) System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. The low pressure portion of the RHR System is designed for 600 psig. An overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent increased risk of core melt. Reference 2 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA. A plant specific review against the NRC criteria for intersystem LOCAs was performed to identify the most risk significant configurations (Ref. 3). Valves identified in this study are listed in the LCO discussion in these Bases. RCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

RCS PIV OPERABILITY protects the low pressure systems attached to the RCS from potential failure due to overpressurization. This protection (that is, RCS PIV OPERABILITY) is provided by the leak tight PIVs.

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken. This LCO only applies to the following PIVs which were determined to be in the most risk significant configurations (Ref. 3):

a. Residual Heat Removal (RHR) System, RHR to loop B accumulator injection line:

Unit 1	SI-6-2
Unit 2	2SI-6-2

b. Safety Injection (SI) System, low pressure SI to upper plenum:

Unit 1	SI-9-3, SI-9-4, SI-9-5, SI-9-6
Unit 2	2SI-9-3, 2SI-9-4, 2SI-9-5, 2SI-9-6

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

Reference 4 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal

BASES

LCO (continued)	pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.
APPLICABILITY	In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.
ACTIONS	The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path when an alternate valve may have degraded the ability of the interconnected system to perform its safety function. <u>A.1 and A.2</u> The flow path must be isolated by two valves. Required Action A.1 is modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the high pressure portion of the system. Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected

BASES

ACTIONS

A.1 and A.2 (continued)

system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

Required Action A.2 specifies that the leaking PIV be restored within limits.

The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this Action and the low probability of a second valve failing during this period.

B.1 and B.2

If leakage cannot be reduced, the system isolated, or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 is required to verify that leakage is below the specified limit and to identify each leaking valve.

The leakage limit of 0.5 gpm per inch of nominal valve diameter up

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1 (continued)

to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition of at least 150 psid.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed at the following times:

a. Every 24 months, a typical refueling cycle;

b. Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months.

The 24 month Frequency is consistent with 10 CFR 50.55a(g) as contained in the Inservice Testing Program, is within the frequency allowed by Reference 4, and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1 (continued)

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures. A differential pressure of at least 150 psid is sufficient to ensure the valves are seated.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

REFERENCES

1. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
2. NUREG-0677, May 1980.
3. Letter from Robert A. Clark, NRC, to L. O. Mayer, NSP, Subject: "Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves," dated April 20, 1981.
4. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section XI.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

AEC GDC 16 (Ref. 1) requires that means be provided for monitoring reactor coolant pressure boundary (RCPB) to detect RCS LEAKAGE. Reference 2 describes methods used for RCS leakage detection.

Leakage detection systems must have the capability to detect significant RCPB degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump A pump run time instrumentation may be used to detect increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} μCi/cc radioactivity for particulate monitoring and of 10^{-6} μCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

One of the containment atmosphere (gaseous and particulate) radiation monitoring channels, R-11 or R-12, normally provides the required monitoring.

BASES

BACKGROUND (continued)

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Humidity measurements can be used as a less sensitive indicator of potential RCS LEAKAGE (Ref. 2).

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed changes in containment sump A pump run time. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

APPLICABLE SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the USAR (Refs. 2 and 3). Multiple instrument locations are utilized, if needed, to ensure that the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into containment is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leakage occur detrimental to the safety of the unit and the public. Containment radionuclide monitoring used for RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii). Containment sump A monitoring used for RCS leakage detection instrumentation satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).
LCO	One method of protecting against large RCS leakage derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation. The LCO is satisfied when monitors of diverse measurement means are available to provide indication of RCS leakage. Thus, the containment sump A monitor (pump run time instrumentation), in combination with a containment radionuclide monitor, provides an acceptable minimum.

BASES (continued)

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is to be ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

The Actions are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment sump and required radionuclide monitors are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

A.1 and A.2

With the required containment sump monitor inoperable, no other form of sampling can provide the equivalent information; however, the containment radionuclide monitor will provide indications of changes in leakage. Together with the radionuclide monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.14.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 3.4.14.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, equilibrium xenon, pressurizer and makeup tank levels, makeup and letdown, and reactor coolant pump seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

BASES

ACTIONS

A.1 and A.2 (continued)

Restoration of the required sump monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

B.1.1, B.1.2, and B.2

When the required containment radionuclide monitoring instrumentation channel is inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.14.1, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment radionuclide monitor.

The 24 hour interval provides periodic information that is adequate to detect leakage. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

C.1 and C.2

If a Required Action of Condition A or B cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating

BASES

ACTIONS

C.1 and C.2 (continued)

experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires the performance of a CHANNEL CHECK of the required containment radionuclide monitor. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.16.2

SR 3.4.16.2 requires the performance of a COT on the required containment radionuclide monitor. The test ensures that the monitor can perform its function in the desired manner. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The test verifies the alarm

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.16.2 (continued)

setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.16.3 and SR 3.4.16.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 24 months is a typical refueling cycle and considers channel reliability.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criterion 16, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. USAR, Section 6.5.

3. USAR, Section 7.5.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident with an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are

BASES

APPLICABLE SAFETY ANALYSES (continued)

conservative in that specific site parameters of the Prairie Island site, such as site boundary location and meteorological conditions, were not considered in this evaluation (Ref. 2).

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.

The SGTR accident analysis (Ref. 3) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature $\geq$ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.

BASES

APPLICABILITY (continued)

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.17-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

Permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 μCi/gm DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4.17-1, accommodates the possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 μCi/gm DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4.17-1 should be minimized since the activity levels allowed by the figure increase the dose at the site boundary following a postulated steam generator tube rupture.

A Note to the ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is

BASES

ACTIONS

A.1 and A.2 (continued)

limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1

With the gross specific activity in excess of the allowed limit, the reactor must be placed in a MODE in which the requirement does not apply. The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in a SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner without challenging plant systems.

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.17-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner without challenging plant systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

SR 3.4.17.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

SR 3.4.17.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.17.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10 CFR 100.11, 1973.

2. Letter from Dominic C. DiIanni, NRC, to L. O. Mayer, NSP, dated December 4, 1981.

3. USAR, Section 14.5.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.18 RCS Loops - Test Exceptions

BASES

BACKGROUND

The primary purpose of this test exception is to provide an exception to LCO 3.4.4, "RCS Loops - MODES 1 and 2," to permit reactor criticality under no flow conditions during certain PHYSICS TESTS (natural circulation demonstration, station blackout, and loss of offsite power) to be performed while at low THERMAL POWER levels. Section XI of 10 CFR 50, Appendix B, requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the power plant as specified in AEC GDC Criterion 1 (Ref. 1).

The key objectives of a test program are to provide assurance that the facility has been adequately designed to validate the analytical models used in the design and analysis, to verify the assumptions used to predict plant response, to provide assurance that installation of equipment at the unit has been accomplished in accordance with the design, and to verify that the operating and emergency procedures are adequate. Testing is performed prior to initial criticality, during startup, and following low power operations.

APPLICABLE SAFETY ANALYSES

Operating the plant without forced convection flow is not bounded by any safety analyses. However, operating experience has demonstrated this exception to be safe under the present applicability.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain

BASES

APPLICABLE SAFETY ANALYSES (continued)	operations by appropriately modifying requirements of other LCOs. A discussion of the criteria for the other LCOs is provided in their respective Bases.
LCO	This LCO provides an exemption to the requirements of LCO 3.4.4. The LCO is provided to allow for the performance of PHYSICS TESTS in MODE 2 (after a refueling), where the core cooling requirements are significantly different than after the core has been operating. Without the LCO, plant operations would be held bound to the normal operating LCOs for reactor coolant loops and circulation (MODES 1 and 2), and the appropriate tests could not be performed. In MODE 2, where core power level is considerably lower and the associated PHYSICS TESTS must be performed, operation is allowed under no flow conditions provided THERMAL POWER is ≤ P-7 and the reactor trip setpoints of the OPERABLE power level channels are set ≤ the allowable value of Table 3.3.1-1, Function 2.b. This ensures, if some problem caused the plant to enter MODE 1 and start increasing plant power, the Reactor Trip System (RTS) would automatically shut it down before power became too high, and thereby prevent violation of fuel design limits. The exemption is allowed even though there are no bounding safety analyses. However, these tests are performed under close supervision during the test program and provide valuable information on the plant's capability to cool down without offsite power available to the reactor coolant pumps.
APPLICABILITY	This LCO is applicable when performing low power PHYSICS TESTS without any forced convection flow. This testing is performed to establish that heat input from nuclear heat does not

BASES

APPLICABILITY (continued)

exceed the natural circulation heat removal capabilities. Therefore, no safety or fuel design limits will be violated as a result of the associated tests.

ACTIONS

A.1

When THERMAL POWER is ≥ the P-7 interlock setpoint, the only acceptable action is to ensure the reactor trip breakers (RTBs) are opened immediately in accordance with Required Action A.1 to prevent operation of the fuel beyond its design limits. Opening the RTBs will shut down the reactor and prevent operation of the fuel outside of its design limits.

SURVEILLANCE REQUIREMENTS

SR 3.4.18.1

Verification that the power level is < the P-7 interlock setpoint will ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. The Frequency of once per hour is adequate to ensure that the power level does not exceed the limit. Plant operations are conducted slowly during the performance of PHYSICS TESTS and monitoring the power level once per hour is sufficient to ensure that the power level does not exceed the limit.

SR 3.4.18.2

The power range and intermediate range neutron channels and the P-7 interlock setpoint must be verified to be OPERABLE and adjusted to the proper value. A COT is performed prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.18.2 (continued) This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.
REFERENCES	1. AEC "General Design Criteria for Nuclear Power Plant Construction," Criterion 1, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply water to the reactor vessel during the blowdown phase of a large break loss of coolant accident (LOCA), to provide inventory to help accomplish the refill and reflood phases that follow thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The reactor coolant inventory is vacating the core during this phase through steam flashing and ejection out through the break. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ejection out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is available to help fill voids in the lower plenum and reactor vessel downcomer, and to help the ongoing reflood of the core with the addition of water.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

BASES

BACKGROUND (continued)

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by a motor operated isolation valve and two check valves in series. The motor operated isolation valves are MV 32071 and MV 32072 (Unit 2 - MV 32174 and MV 32175) (Westinghouse valve numbers 8800A and 8800B respectively for both units).

The accumulator size, water volume, and nitrogen cover pressure are selected so that one of the two accumulators is sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that one accumulator is adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 1). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow. In the early stages of a large break LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting large break LOCA is a double ended guillotine break at the discharge of the reactor coolant pump. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for safety injection (SI) signal generation, the diesels starting and the pumps being loaded and delivering full flow. The SI signal generation occurs approximately 2 seconds into the transient. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and safety injection pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the safety injection pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 will be met following a LOCA:

a. The calculated peak fuel element cladding temperature is below the requirement of 2200°F;

b. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limits of 17% are not exceeded during or after quenching;

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. The amount of hydrogen generated by fuel element cladding that reacts chemically with water or steam does not exceed an amount corresponding to interaction of 1% of the total amount of Zircaloy in the reactor; and

d. The core remains amenable to cooling during and after the break.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For the large break LOCA analyses, a nominal contained accumulator water volume of 1270 cubic feet is used based on minimum and maximum volumes of 1250 cubic feet (25% indicated level) and 1290 cubic feet (91% indicated level).

The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. Prairie Island is a two loop plant with Upper Plenum Injection (UPI) LOCA analyses. For UPI plant small breaks, a decrease in water volume is a peak clad temperature penalty; thus, a minimum contained water volume is assumed. Both large and small break analyses use a nominal accumulator line water volume from the accumulator to the check valve.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent

BASES

APPLICABLE SAFETY ANALYSES (continued)

reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. For conservatism, the accumulators are not considered in the boron build up analyses since their inclusion would dilute the sump concentration.

The small break LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The large break analyses utilize the nominal nitrogen cover pressure as per approved methods (Ref. 1). The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 1 and 2).

The accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Two accumulators are required to ensure that 100% of the contents of one accumulator will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than one accumulator is injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 could be violated.

For an accumulator to be considered OPERABLE, the motor-operated isolation valve must be fully open, power removed above 2000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

BASES (continued)

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 limit of 2200°F.

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood since the accumulator water volume is very small when compared to RCS and RWST inventory. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current

BASES

ACTIONS

A.1 (continued)

analysis techniques demonstrate that the accumulators are not expected to discharge following a large main steam line break. Even if they do discharge, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 24 hours. In this Condition, the required contents of one accumulator cannot be assumed to reach the core during a LOCA.

Due to the severity of the consequences should a LOCA occur in these conditions, the 24 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions. The 24 hours allowed to restore an inoperable accumulator to OPERABLE status were justified in Reference 3.

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the

BASES

ACTIONS

C.1 and C.1 (continued)

plant must be brought to MODE 3 within 6 hours and RCS pressure reduced to ≤ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If both accumulators are inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Each accumulator motor operated valve should be verified to be fully open every 12 hours. Use of control board indication (position monitor lights and alarms) for valve position is an acceptable verification. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed or not fully open valve could result in not meeting accident analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.2 and SR 3.5.1.3

Every 12 hours, borated water volume and nitrogen cover pressure are verified for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage.

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator when the RCS pressure is ≥ 2000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only one accumulator would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when RCS pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns.

BASES (continued)

REFERENCES

1. USAR, Section 14.
2. USAR, Section 6.2.
3. WCAP-15049-A, Revision 1, "Risk-Informed Evaluation of an Extension to Accumulator Completion Times," April 1999.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS-Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Loss of secondary coolant accident, including uncontrolled steam release; and

c. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are two phases of ECCS operation: injection and recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs and reactor vessel upper plenum. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sump has enough water to supply the required net positive suction head to the RHR pumps, suction is switched to containment Sump B for recirculation. When post accident RCS pressure drops below the RHR pump shutoff head, the RHR flow is directed into the reactor vessel upper plenum to reduce the boiling in the top of the core and any resulting boron precipitation.

BASES

BACKGROUND (continued)

The ECCS consists of two separate subsystems: safety injection (SI) and residual heat removal (RHR). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the RHR pumps, RHR heat exchangers, and the SI pumps. Both subsystems consist of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from opposite trains to achieve the required 100% flow to the core if necessary due to individual component inoperability.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps. Separate piping supplies each subsystem. The discharge from each SI pump divides and feeds an injection line to each of the RCS cold legs. Throttle valves are set to balance the flow to the RCS. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs. The discharge from each RHR pump divides and feeds an injection line to the reactor vessel upper plenum.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the steam generators provide core cooling until the RCS pressure decreases below the SI pump shutoff head.

BASES

BACKGROUND (continued)

During the recirculation phase of LOCA recovery, RHR pump suction is manually transferred to the containment sump. Initially, recirculation is through the same paths as the injection phase. The RHR pumps provide flow to the reactor vessel upper plenum. If the RCS pressure limits RHR flow, then the RHR pumps supply the SI pumps which provide flow to the cold legs.

The SI subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCOs 3.4.12, "Low Temperature Overpressure Protection (LTOP) > Safety Injection (SI) Pump Disable Temperature," and 3.4.13, "Low Temperature Overpressure Protection (LTOP) ≤ Safety Injection (SI) Pump Disable Temperature," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguards loads is accomplished in a programmed time sequence. If offsite power is available, the safeguards loads start immediately in the programmed sequence. If offsite power is not available, the safeguards buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguards loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the length of time before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet AEC GDC 44 (Ref. 1).

BASES (continued)

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptancecriteria for the ECCS, established by 10 CFR 50.46, will be met following a LOCA:

a. The calculated peak fuel element cladding temperature is below the requirement of 2200°F;

b. The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limits of 17% are not exceeded during or after quenching;

c. The amount of hydrogen generated by fuel element cladding that reacts chemically with water or steam does not exceed an amount corresponding to interaction of 1% of the total amount of Zircaloy in the reactor;

d. The core remains amenable to cooling during and after the break; and

e. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

The LCO also limits the potential for a post trip return to power following an MSLB event.

Both ECCS subsystems are taken credit for in a large break LOCA event at full power (Refs. 2 and 3). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The SI pumps are credited in small break LOCA, SGTR and MSLB events. The small break LOCA event establishes the flow and discharge head at the design point for the pumps. The SI pump head and flow characteristics also meet SGTR and MSLB requirements. The

BASES

APPLICABLE SAFETY ANALYSES (continued)

OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of a single ECCS train; and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected by the SI pumps into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core. The RHR pumps inject directly into the reactor vessel by upper plenum injection when the RCS pressure is less than the RHR pump shutoff head. The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 2 and 3). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the SI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For smaller LOCAs, the SI pumps deliver sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents

BASES

LCO (continued)

In MODES 1, 2, and 3, an ECCS train consists of an SI subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and RHR capable of being transferred to take suction from containment Sump B.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the two cold leg injection nozzles and the reactor vessel upper plenum. In the long term, this flow path may be switched to take its supply from the containment sump and to supply its flow to the RCS cold legs or directly into the reactor vessel upper plenum.

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

Manual valves that could, if improperly positioned, reduce injection flow below that assumed for accident analyses, are blocked and tagged or locked in the proper position for injection. Changes in valve position must be under direct administrative control.

A block is a device that can be unclipped or unsnapped to allow a status change of the component to which it is applied. A lock is a device that must be unlocked, destroyed or mechanically removed (such as a cap or blank) to allow a status change of the component to which it is applied.

As indicated in the LCO Note, the SI flow paths may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.15.1. The flow path is readily restorable from the control room.

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The SI pump performance requirements are based on a small break LOCA and meet required parameters for mitigation of a secondary side loss of fluid accident. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

This LCO is only applicable in MODE 3 and above.

In MODES 4, 5, and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. MODE 4 core cooling requirements are addressed by LCO 3.5.3, "ECCS-Shutdown," and LCO 3.4.6, "RCS Loops-MODE 4." Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level."

ACTIONS

A.1

With one or more trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 4) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or required supporting systems are not available.

BASES

ACTIONS

A.1 (continued)

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 4) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 5 describes situations in which one component, such as an RHR crossover valve, can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to

BASES

ACTIONS

B.1 and B.2 (continued)

a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

Condition A is applicable with one or more trains inoperable. The allowed Completion Time is based on the assumption that at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available. With less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the facility is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Use of control board indication for valve position is an acceptable verification. Misalignment of these valves could render one or both ECCS trains inoperable. These valves are secured in position by physically locking the motor control center supply breakers in the off position with the valve position monitor lights OPERABLE to assure that they cannot change position as a result of an active failure or be inadvertently misaligned. Verification of the valve breakers is performed by SR 3.5.2.3.

A 12 hour Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned valve is unlikely.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing (A seal is a device that must be destroyed to allow a status change of the component to which it is applied). A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

Verification every 31 days that the motor control center supply breakers are physically locked in the off position for each valve specified in SR 3.5.2.1 ensures that an active failure could not result in an undetected misposition of a valve. Since power is removed under administrative control and valve position is verified every 12 hours, the 31 day Frequency will provide adequate assurance that power is removed.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at a single point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is within the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This test is met when control board indications and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing, the appropriate pump breakers have opened and closed, and all automatic valves have been placed in the proper position required to establish a safety injection flow path to the reactor coolant system.

This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.5 and SR 3.5.2.6 (continued)

Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of Engineered Safety Feature (ESF) Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.7

Surveillance Requirements on ECCS throttle valves provide assurance that proper ECCS flows are maintained in the event of a LOCA. Proper flow resistance and pressure drop in the piping system to each injection point in the SI System is necessary to: 1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration; 2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS LOCA analyses; and 3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS LOCA analyses. The 24 month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6.

SR 3.5.2.8

Periodic inspections of the containment sump suction inlet to the RHR System ensure that it is unrestricted and stays in proper operating condition. The 24 month Frequency allows this Surveillance to be performed under the conditions that apply during a plant outage. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

BASES (continued)

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criterion 44, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. USAR, Section 6.2.

3. USAR, Section 14.

4. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

5. IE Information Notice No. 87-01.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS – Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS-Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: safety injection (SI) and residual heat removal (RHR).

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) or containment Sump B can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

Due to the lower heat generation rate associated with operation in MODE 4 it has been judged that the full power licensing analyses described in the Applicable Safety Analyses section of Bases 3.5.2 would bound the consequences of a Design Basis Accident (DBA) in MODE 4. It is also recognized that due to the lower pressure and temperatures in the RCS, the probability of occurrence of a DBA is reduced. Therefore, the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic SI actuations are not available. Since the RHR System may be aligned to provide normal shutdown cooling, time may be required for manual alignment of ECCS equipment. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA. Therefore, only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered for this LCO due to the time available for operators to respond to an accident.

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

In MODE 4, an ECCS train consists of an SI subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the cold leg injection nozzles and reactor vessel upper plenum. In the long term, this flow path may be switched to take its supply from the containment sump.

This LCO is modified by a Note that allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F and both RCS cold leg temperatures above the SI pump disable temperature specified in the PTLR, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODE 4 when any RCS cold leg temperature is ≤ the SI pump disable temperature, and MODES 5 and 6, plant conditions are such

BASES

APPLICABILITY (continued)

that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 4 when any RCS cold leg temperature is ≤ the SI pump disable temperature are addressed by LCO 3.4.6, "RCS Loops-MODE 4." Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level."

ACTIONS

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

BASES

ACTIONS (continued)

B.1

With no ECCS SI subsystem OPERABLE (neither train), due to the inoperability of the SI pump or flow path from the RWST, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one SI subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required.

C.1

When the Required Actions of Conditions B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

The applicable references from Bases 3.5.2 apply.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling pool during refueling, and to the ECCS and the Containment Spray System during accident conditions.

The RWST supplies both trains of the ECCS and the Containment Spray System during the injection phase of a loss of coolant accident (LOCA). A motor operated isolation valve is provided in each header to isolate the RWST from the ECCS once the system has been transferred to the recirculation mode. The recirculation mode is entered when RHR pump suction is transferred to the containment sump following receipt of the RWST-Low Low Level alarm. Use of a single RWST to supply both trains of the ECCS and Containment Spray System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with initiation of Design Basis Events.

The RWST is located in the Auxiliary Building which maintains the tank temperature above freezing and therefore maintains the boron soluble (Ref. 1).

During normal operation in MODES 1, 2, and 3, the safety injection (SI), residual heat removal (RHR), and Containment Spray (CS) pumps are aligned to take suction from the RWST.

The ECCS and CS pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions. The recirculation lines for the RHR pumps are directed from the discharge of the pumps to the pump suction. The recirculation lines for the SI and CS pumps are directed back to the RWST.

BASES

BACKGROUND (continued)

When the suction for the ECCS pumps is transferred to the containment sump, the RWST and SI pump recirculation flow paths must be isolated to prevent a release of the containment sump contents to the RWST, which could result in a release of contaminants to the Auxiliary Building atmosphere and the eventual loss of suction head for the ECCS pumps.

This LCO ensures that:

a. The RWST contains sufficient borated water to support the ECCS during the injection phase;

b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS pumps at the time of transfer to the recirculation mode of cooling; and

c. The reactor remains subcritical following a LOCA.

Insufficient water in the RWST could result in inadequate net positive suction head (NPSH) for the RHR pumps when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SHUTDOWN MARGIN (SDM) or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS Operating"; B 3.5.3, "ECCS-Shutdown"; and B 3.6.5, "Containment Spray and Cooling Systems." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RWST must also meet volume and boron concentration requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is determined by the volume of water required in the containment sump to provide the necessary NPSH for the RHR pumps at the time of switchover to recirculation. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. The maximum boron concentration is an explicit assumption in the evaluation of chemical effects resulting from the operation of the CS System. Temperatures above freezing in the RWST in combination with the maximum boron concentration ensure that the boron will remain soluble while in the RWST.

For a large break LOCA analysis, the minimum water volume limit of 200,000 gallons (68% of indicated level) and the lower boron concentration limit of 2600 ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of 3500 ppm is used in calculations which verify boron precipitation does not occur in the core following a LOCA. The upper limit on boron concentration is also used in containment sump chemistry calculations to assure that post-LOCA pH is within acceptable limits.

The RWST satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and

BASES

LCO (continued)

to ensure adequate level in the containment sump to support ECCS pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume and boron concentration limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and Containment Spray System OPERABILITY requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level."

ACTIONS

A.1

With RWST boron concentration not within limits, it must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST boron concentration to within limits was developed considering the time required to change the boron concentration and the fact that the contents of the tank are still available for injection.

B.1

With the RWST water volume not within limits, it must be restored to OPERABLE status within 1 hour.

BASES

ACTIONS

B.1 (continued)

In this Condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS pump operation on recirculation. Since the RWST volume is normally stable and the RWST is located in the Auxiliary Building which provides leak detection capability, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

BASES

SURVEILLANCE REQUIREMENTS (continued)	<u>SR 3.5.4.2</u> The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.
REFERENCES	1. USAR, Section 6 and Section 14.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment is a free standing steel pressure vessel surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low leakage steel shell designed to contain radioactive material that may be released from the reactor core following a design basis Loss of Coolant Accident (LOCA). Additionally, the containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with a hemispherical dome and ellipsoidal bottom, completely enclosed by a reinforced concrete shield building. A 5 ft wide annular space exists between the walls of the steel containment vessel and the concrete shield building and 7 ft clearance exists between the roofs of the containment vessel and shield building to permit inservice inspection and collection of containment outleakage.

Containment piping penetration assemblies provide for the passage of process, service, sampling and instrumentation pipelines into the containment vessel while maintaining containment OPERABILITY. The shield building provides shielding and allows controlled release of the annulus atmosphere under accident conditions, as well as environmental missile protection for the containment vessel and the Nuclear Steam Supply System.

The inner steel containment and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50 Appendix J, Option B (Ref. 1), as modified by approved exemptions.

BASES

BACKGROUND (continued)

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE automatic containment isolation system, or

   2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks"; and

c. All equipment hatches are closed and sealed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting Design Basis Accident (DBA) without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The reactor containment vessel, including the penetrations, is designed for low leakage to minimize the consequences (dose) to the general public during a DBA. The maximum allowable containment leakage rate is an input to the dose analyses. In the Updated Safety Analysis Report (USAR), the maximum allowable containment leakage used in the large break

BASES

APPLICABLE SAFETY ANALYSES (continued)

LOCA dose analysis was 2.5 weight percent per day. In the SER, the AEC concluded that a maximum containment leakage of 0.5 weight percent per day was acceptable. This formed the basis for the original plant Technical Specification leakage limit of 0.5 weight percent per day. Subsequently, it was concluded that the Shield Building leakage was higher than anticipated which increased the calculated dose. With the higher Shield Building leakage, in order to reduce the calculated dose, the maximum allowable containment leakage was reduced to 0.25 weight percent per day (Ref. 2). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined for Prairie Island in the Containment Leakage Rate Testing Program as L_a: the maximum allowable containment leakage rate at the containment design maximum internal pressure (P_a). The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.25% per day in the safety analysis at P_a = 46.0 psig (Ref. 2).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage rate limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2), purge valves with resilient seals, and secondary bypass

BASES

LCO (continued)	leakage (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a or exceeding the total maximum allowable secondary containment bypass leakage rates.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment OPERABLE during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable,

BASES

ACTIONS

B.1 and B.2 (continued)

based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock, secondary containment (shield building and auxiliary building special ventilation zone) bypass leakage path and inservice purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be $\leq 0.6\ L_a$ for combined Type B and C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria are based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.2

Verifying that the maximum temperature differential between average containment and annulus air temperatures is less than or equal to 44 °F ensures that containment operation remains within the limits assumed for the containment analyses. Plant operating experience demonstrates that this limit can only be approached when the plant is in MODES 5 and 6. Requiring this temperature differential to be verified prior to entering MODE 4 from MODE 5 provides assurance this parameter is within acceptable limits prior to establishing conditions requiring containment integrity.

SR 3.6.1.3

Verifying that the minimum containment shell temperature is met ensures that adequate margin above NDTT exists. Plant operating experience demonstrates that this limit can only be approached when the plant is in MODES 5 and 6. Requiring containment shell temperature to be verified prior to entering MODE 4 from MODE 5 provides assurance that the shell temperature is above NDTT prior to establishing conditions requiring containment integrity.

REFERENCES

1. 10 CFR 50, Appendix J.

2. USAR, Section 14.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder, 10 ft in diameter, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a design basis accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each air lock is provided with limit switches on both doors that provide control room indication of door position. Additionally, control room indication alerts the operator whenever both air lock doors are open which indicates the interlock mechanism is defeated.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analyses.

BASES (continued)

APPLICABLE SAFETY ANALYSES	The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Ref. 1). The Loss of Coolant Accident (LOCA) dose analysis bounds the rod ejection accident releases. In the LOCA analysis, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The assumed containment leakage rate is 0.25% of containment air weight per day (Ref. 1). This leakage rate is defined at Prairie Island in the Containment Leakage Rate Testing Program as L_a, the maximum allowable containment leakage rate at the containment internal design pressure P_a = 46.0 psig. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks. The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event. Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the 10CFR50, Appendix J, Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events.

BASES

LCO (continued)	Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment. Normal entry into or exit from containment does not render the air lock inoperable.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."
ACTIONS	The ACTIONS are modified by three Notes. The first Note allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. For repairs to the inner door, it is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from inside the air lock between the two doors then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

BASES

ACTIONS (continued)

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour and may consist of verifying the control board alarm status for the air lock doors. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A, only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required on a periodic basis to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment is entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

BASES

ACTIONS (continued)

B.1, B.2, and B.3

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B (e.g., both doors of an air lock are inoperable), Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation per LCO 3.6.1 is acceptable, since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal

BASES

ACTIONS

C.1, C.2, and C.3 (continued)

test or if the overall air lock leakage is not within the limits of SR 3.6.2.1. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits due to the large margin between the air lock leakage and the containment overall leakage acceptance criteria.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour. Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria which are applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2 (continued)

outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the potential for loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. The 24 month Frequency for the interlock is justified based on generic operating experience. The 24 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during the use of the airlock.

REFERENCES

1. USAR, Chapter 14.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated power operated valves secured in their closed position, check valves with flow through the valve secured, blind flanges, and closed systems are considered passive devices. Automatic valves designed to close without operator action following an accident are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system which means it penetrates primary containment, is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere, and has a low probability of being ruptured by an accident (Refs. 1 and 2). These barriers (typically containment isolation valves) make up the Containment Isolation System.

The Containment Isolation System is designed to provide isolation capability following a design basis accident (DBA) for fluid lines which penetrate containment. Major non-essential lines (i.e., fluid systems which do not perform an immediate accident mitigation function) which penetrate containment, except for main steam lines, are either automatically isolated following an accident or are normally maintained closed in MODES 1, 2, 3, and 4. Automatic containment isolation valves are designed to close on a containment isolation signal which is generated by either an automatic safety injection (SI) signal or by manual actuation. The Containment Isolation System can also isolate essential lines at the discretion of

BASES

BACKGROUND (continued)

the operators depending on the accident progression and mitigation requirements.

Upon receipt of a containment pressure High-High signal, both main steam isolation valves close which also causes the instrument air line to containment to isolate if a containment isolation signal is also present. In addition to the isolation signals listed above, the containment purge and inservice purge supply and exhaust line valves and dampers receive isolation signals on a safety injection signal, a containment high radiation condition, a manual containment isolation actuation and manual containment spray initiation. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the outside environment in the event of a release of fission product radioactivity to the containment atmosphere resulting from a DBA.

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analyses. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analyses will be maintained.

BASES

BACKGROUND (continued)

In addition to the normal fluid systems which penetrate containment, two systems which can provide direct access from inside containment to the outside environment are described below.

Containment Purge System (36 inch purge valves)

The Containment Purge System operates to supply outside air into the containment for ventilation and cooling or heating and may also be used to reduce the concentration of noble gases within containment prior to and during personnel access in MODES 5 and 6. The supply and exhaust lines each contain one isolation valve, one isolation damper and a blind flange. The 36 inch purge valves and dampers are not tested to verify their leakage rate is within the acceptance criteria of the Containment Leakage Rate Testing Program. Therefore, blind flanges are installed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

Inservice Purge System (18 inch purge valves)

The Inservice Purge System operates to:

a. Reduce the concentration of noble gases within containment prior to and during personnel access; and

b. Provide low volume normal purge and ventilation.

Two containment automatic isolation valves and an automatic shield building ventilation damper are provided on each supply and exhaust line. The supply and exhaust lines are designed to have blind flanges installed where the lines pass through the shield

BASES

BACKGROUND

Inservice Purge System (18 inch purge valves) (continued)

building annulus. Normally, during MODES 1, 2, 3, and 4 the blind flanges provide the containment penetration isolation function. When ventilation of containment is required in MODES 1, 2, 3, and 4, the valves will be leak tested, and the blind flanges removed and replaced with a spool piece. Prior to system use, the automatic isolation valves and dampers are verified to be OPERABLE and a debris screen is installed on each line preventing foreign material from inhibiting the proper closing of the valves. When purge of containment is completed and inservice purge system operation is no longer required, the system is returned to its normal operating configuration with the spool pieces removed. The blind flanges are installed on penetrations 42B and 43A (52 and 53 in Unit 2) and tested to meet the acceptance criteria of the Containment Leakage Rate Testing Program.

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material to the containment atmosphere are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 3). In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves are minimized. The safety analyses assume that the 36 inch purge lines are blind flanged at event initiation.

BASES

APPLICABLE SAFETY ANALYSES (continued)

In calculation of control room and offsite doses following a LOCA, the accident analyses assume that 25% of the equilibrium iodine inventory and 100% of the equilibrium noble gas inventory developed from maximum full power operation of the core is immediately available for leakage from containment (Ref. 3). The containment is assumed to leak at the maximum allowable leakage rate, L_a, for the first 24 hours of the accident and at 50% of this leakage rate for the remaining duration of the accident.

The containment penetration isolation valves ensure that the containment leakage rate remains below L_a by automatically isolating penetrations that do not serve post accident functions and providing isolation capability for penetrations associated with Engineered Safety Features. The maximum isolation time for automatic containment isolation valves is 60 seconds. This isolation time is based on engineering judgement since the control room and offsite dose calculations are performed assuming that leakage from containment begins immediately following the accident with no credit for transport time or radioactive decay. The 60 second isolation time takes into consideration the time required to drain piping of fluid which can provide an initial containment isolation before the containment isolation valves are required to close and the conservative assumptions with respect to core damage occurring immediately following the accident.

The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

The containment inservice purge valves have been analyzed to demonstrate they are capable of closing during the design basis LOCA (Ref. 2). During plant operation, the containment inservice purge lines are normally blank flanged and the valves are not relied upon as penetration isolation devices.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Containment isolation also isolates the Reactor Coolant System (RCS) to prevent the release of radioactive material. However, RCS isolation, not isolation of containment, is required for events which result in failed fuel and do not breach the integrity of the RCS (e.g., reactor coolant pump locked rotor). The isolation of containment following these events also isolates the RCS from all non-essential systems to prevent the release of radioactive material outside the RCS. The containment isolation time requirements for these events are bounded by those for the LOCA.

The Containment Isolation System is designed to provide two boundaries for each penetration such that no single credible failure or malfunction (expected fault condition) occurring in any active system component can result in loss of isolation or intolerable leakage in compliance with the AEC GDC 53, "Containment Isolation Valves," (Ref. 4).

The containment isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

The containment isolation devices covered by this LCO consist of isolation valves (manual valves, check valves, air operated valves, and motor operated valves), pipe and end caps, closed systems, and blind flanges.

BASES

LCO (continued)

Vent and drain valves located between two isolation devices are also containment isolation devices. Test connections located between two isolation valves are similar to vent and drain lines except that no valve may exist in the test line. A cap or blind flange, as applicable, must be installed on these vent, drain and test lines. A cap or blind flange installed on these lines make them "otherwise secured" for SR considerations.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 36 inch purge valves must be blind flanged in MODES 1, 2, 3, and 4. The valves covered by this LCO are listed in Reference 2.

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic power operated valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 2.

Inservice purge valves with resilient seals and secondary containment (shield building and auxiliary building special ventilation zone) bypass valves must meet additional leakage rate requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing.

This LCO provides assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by four Notes. The first Note allows penetration flow paths, except for 36 inch containment purge system penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the blind flanges on the containment purge system lines during plant operation, the penetration flow path containing these flanges may not be opened under administrative controls.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

BASES

ACTIONS (continued)

In the event containment isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable, except for inservice purge penetrations or secondary containment bypass leakage not within limit, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated or mechanically blocked power operated containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not

BASES

ACTIONS

A.1 and A.2 (continued)

require any testing or device manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides the appropriate actions.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices once they have been verified to be in the proper position, is small.

BASES

ACTIONS (continued)

B.1

With two containment isolation valves in one or more penetration flow paths inoperable, except for inservice purge penetration or secondary containment bypass leakage not within limits, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

BASES

ACTIONS (continued)

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated power operated valve, a closed manual valve, and a blind flange. With the exception of the chemical and volume control system (CVCS), a check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. This required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements defined in Reference 2. This Note is

BASES

ACTIONS

C.1 and C.2 (continued)

necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

Required Action C.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

D.1

With the secondary containment bypass leakage rate (SR 3.6.3.8) or inservice purge valve(s) (SR 3.6.3.6) leakage is not within limit, the assumptions of the safety analyses are not met. Therefore, the leakage must be restored to within limit within 4 hours. Restoration can be accomplished by isolating the penetration(s) that caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange.

BASES

ACTIONS

D.1 (continued)

When a penetration is isolated the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 4 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration(s) and the relative importance of secondary containment bypass leakage and containment purge penetration valve(s) leakage to the overall containment function.

E.1

In the event containment purge blind flange leakage (SR 3.6.3.1) or inservice purge blind flange leakage (SR 3.6.3.2) are not within limits, the leakage rate must be restored within 1 hour to assure containment leakage rates are met. If containment purge blind flange leakage rate or inservice purge blind flange leakage rate limits are not met, it could be due to the blind flange not installed or improperly installed. If containment purge blind flange leakage rate or inservice purge blind flange leakage rate limits are not met, it could be due to the blind flange not installed or improperly installed. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment OPERABLE during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when blind flange leakage exceeds its limits is minimal.

BASES

ACTIONS (continued)

F.1 and F.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each 36 inch containment purge system penetration is required to be blind flanged when the plant is in MODES 1, 2, 3, and 4. This Surveillance is designed to ensure that the blind flange is installed prior to entering MODE 4 from MODE 5.

SR 3.6.3.2

This SR ensures that the 18 inch containment inservice purge penetrations are blind flanged after each use of the system. Since the inservice purge penetration blind flanges are part of the containment boundary, they are required to meet the Containment Leakage Rate Testing Program acceptance criteria required by SR 3.6.1.1 as required by this SR.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment manual valves and blind flanges outside containment and capable of being mispositioned are in the correct position. Since verification of manual valve and blind flange position for containment isolation valves outside containment is relatively easy, the 92 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For containment isolation manual valves and blind flanges inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

This Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these containment isolation valves or blind flanges, once they have been verified to be in their proper position, is small.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.5

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.6

Since PI only uses the containment inservice purge system infrequently for short periods of time, this SR must be performed prior to each use of the system when containment integrity is required to assure that the valve leakage rate is within an acceptable value.

SR 3.6.3.7

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.8

This SR ensures that the combined leakage rate of all secondary containment (shield building and auxiliary building special ventilation zone) bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. The acceptance criteria and Frequency are provided by the Containment Leakage Rate Testing Program.

Bypass leakage is considered part of L_a.

REFERENCES

1. 10 CFR 50, Appendix A.
2. USAR, Section 5.2.
3. USAR, Section 14.
4. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criterion 53, issued for comment, July 10, 1967, as referenced in USAR Section 1.2.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment analyses. Should operation occur outside this limit coincident with a LOCA or SLB, post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the LOCA and SLB analyses to establish the maximum peak containment internal pressure. The limiting events considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure models. The worst case SLB generates larger mass and energy release than the worst case LOCA. Thus, the SLB event bounds the LOCA event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 16.7 psia (2.0 psig). This resulted in a maximum peak pressure from a SLB of less than 46 psig. The containment analyses show that the maximum peak calculated containment pressure results from the SLB. The maximum containment pressure resulting from the SLB does not exceed the containment design maximum internal pressure, 46 psig.

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a LOCA or SLB, the resultant peak containment accident pressure will remain below the containment design maximum internal pressure.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 8 hours. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is greater than the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour. However, due to the large containment free volume and limited size of the post-LOCA vent system, 8 hours is allowed to restore containment pressure to within limits. This is justified by the low probability of a DBA during this time period.

BASES

ACTIONS (continued)	<u>B.1 and B.2</u> If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	<u>SR 3.6.4.1</u> Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.
REFERENCES	1. USAR, Section 14.5.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Spray and Cooling Systems

BASES

BACKGROUND

The Containment Spray and Containment Cooling Systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a design basis accident (DBA), to within limits. The Containment Spray and Containment Cooling Systems are designed, as described in the USAR, to meet the requirements of AEC GDC 37, "Engineered Safety Features Basis for Design," GDC 38, "Reliability and Testing of Engineered Safety Features," GDC 41, "Engineered Safety Features Performance Capability," GDC 42, "Engineered Safety Features Components Capability," GDC 49, "Containment Design Basis," GDC 52, "Containment Heat Removal Systems," GDC 58, "Inspection of Containment Pressure-Reducing Systems," GDC 59, "Testing of Containment Pressure-Reducing Systems," GDC 60, "Testing of Containment Spray Systems," and GDC 61, "Testing of Operational Sequence of Containment Pressure-Reducing Systems," (Ref. 1).

The Containment Cooling System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained.

Containment Spray System

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases.

BASES

BACKGROUND

Containment Spray System (continued)

Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of a DBA.

The Containment Spray System provides a spray of cold borated water mixed with sodium hydroxide (NaOH) from the spray additive tank into the upper regions of containment to reduce the containment pressure and temperature and to remove fission products from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System. Each train of the Containment Spray System provides adequate spray coverage to provide 100% of the Containment Spray System design requirements for containment heat removal.

The Spray Additive System mixes an NaOH solution into the spray. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the containment sump. Controlling the alkaline pH of the containment sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The Containment Spray System is actuated either automatically by a containment High-High pressure signal or manually. An automatic actuation signal opens the containment spray pump discharge valves, opens the Spray Additive System valves, starts the two containment spray pumps, and begins injection. A manual actuation of the Containment Spray System requires the operator to simultaneously actuate two separate switches on the main control board to begin the

BASES

BACKGROUND Containment Spray System (continued)

same sequence. The spray injection continues until containment pressure is reduced to less than 20 psig or an RWST level Low-Low alarm is received. When one of these conditions is reached, containment spray is manually terminated.

Due to the nature of the containment spray system, most functional tests are performed with the isolation valves in the spray supply lines at containment and the spray additive tank isolation valves blocked closed. The tests are considered satisfactory if visual observations indicate all components have operated satisfactorily.

Containment Cooling System

Two trains of containment cooling, each of sufficient capacity to supply 100% of the Containment Cooling System design cooling requirements, are provided. Each train of two fan coil units is normally supplied with chilled water during summer operation or cooling water from separate trains of the Cooling Water System (CL) for winter or emergency operation. Air is drawn into the coolers through the fan and discharged to the containment atmosphere including various compartments (e.g., steam generator and pressurizer compartments).

During normal operation, all four fan coil units are operating. The fans may be operated at high or low speed with chilled water (summer operation) or CL water supplied to the cooling coils. The Containment Cooling System is designed to limit the ambient containment air temperature during normal unit operation to less than 120° F. This temperature limitation ensures that the containment temperature does not exceed the initial temperature conditions assumed for the DBAs.

BASES

BACKGROUND

Containment Cooling System (continued)

In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically in slow speed if not already running. If running in high speed, the fans automatically shift to slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher mass atmosphere. The temperature of the cooling water is an important factor in the heat removal capability of the fan coil units.

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Cooling System limit the temperature and pressure that could be experienced following a loss of coolant accident (LOCA) or steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. These events are not assumed to occur simultaneously or consecutively. These postulated events are analyzed with regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train of the Containment Spray System and Containment Cooling System being rendered inoperable.

The analyses and evaluations show that under the worst case scenario, the highest peak containment pressure is less than 46 psig. The analyses show that the peak containment temperature meets the intent of the design basis. The analyses and evaluations assume a conservative unit specific power level for the accident under consideration (LOCA or SLB), one containment spray train and one containment cooling train operating, and conservative initial (pre-accident) containment pressure of 2.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

BASES

APPLICABLE SAFETY ANALYSES (continued)

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K.

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a containment pressure reduction associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.8.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-High pressure setpoint to achieving full flow through the containment spray nozzles.

The analyses of the Main Steam Line Break (MSLB) and LOCA incorporated delays in Containment Spray actuation to account for load restoration, discharge valve opening, containment spray pump windup, and spray line filling (Ref. 3).

Containment cooling train performance for post accident conditions is given in Reference 4. The result of the analyses is that one train of containment cooling with one train of containment spray can provide 100% of the required peak cooling capacity during post accident conditions. The train post accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference 5.

The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with receiving a safety injection (SI) signal to achieving full

BASES

APPLICABLE SAFETY ANALYSES (continued)

Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time incorporates delays to account for load restoration and motor windup (Ref. 3).

The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a LOCA or SLB, a minimum of one containment cooling train and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits (Ref. 4). Additionally, one containment spray train is also required to remove iodine from the containment atmosphere and thereby maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and two containment cooling trains must be OPERABLE. Therefore, in the event of an accident, at least one train in each system operates, assuming the worst case single active failure occurs.

Each Containment Spray System includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon a containment spray actuation signal. Manual valves in this system that could, if improperly positioned, reduce the spray flow below that assumed for accident analysis, are blocked and tagged in the proper position and maintained under administrative control. Containment Spray System motor operated valves, MV-32096 and MV-32097 (Unit 1), and MV-32108 and MV-32109 (Unit 2) are closed with the motor control center supply breakers in the off position.

Each Containment Cooling System typically includes cooling coils, dampers, fans, and controls to ensure an OPERABLE flow path.

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a LOCA or SLB could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains and containment cooling trains.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray System and the Containment Cooling System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the other Containment Spray train, reasonable time for repairs, and low probability of a LOCA or SLB occurring during this period.

The 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, "Completion Times," for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

BASES

ACTIONS (continued)

B.1 and B.2

If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time for attempting restoration of the containment spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

With one of the containment cooling trains inoperable, the inoperable containment cooling train must be restored to OPERABLE status within 7 days. In this degraded condition the remaining OPERABLE containment spray and cooling trains provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs. The 7 day Completion Time was developed taking into account the heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action C.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

BASES

ACTIONS (continued)

D.1 and D.2

If the Required Action and associated Completion Time of Condition C of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment (there are no valves inside containment) and capable of potentially being mispositioned are in the correct position.

SR 3.6.5.2

Operating each containment cooling train fan coil unit on low motor speed for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. Motor current is measured and compared to the nominal current expected for the

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.2 (continued)

test condition. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the fan coil units and controls, the two train redundancy available, and the low probability of significant degradation of the containment cooling train occurring between Surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.5.3

Verifying that each containment cooling train cooling water flow rate to each fan coil unit is ≥ 900 gpm provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 4). Terminal temperatures of each fan coil unit are also observed. This test includes verifying operation of all essential features including low motor speed, cooling water valves and normal ventilation system dampers. The 24 month Frequency is based on; the need to perform these Surveillances under the conditions that apply during a plant outage; the known reliability of the Cooling Water System; the two train redundancy available; and, the low probability of a significant degradation of flow occurring between Surveillances.

SR 3.6.5.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code. Since the

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.4 (continued)

containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

SR 3.6.5.5 and SR 3.6.5.6

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment High-High pressure signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. To prevent inadvertent spray in containment, containment spray pump testing with a simulated actuation signal will be performed with the isolation valves in the spray supply lines at the containment and the spray additive tank isolation valves blocked closed. These tests will be considered satisfactory if visual observations indicate all components have operated satisfactorily. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.7

This SR requires verification that each containment cooling train actuates upon receipt of an actual or simulated safety injection signal. The 24 month Frequency is based on engineering judgment. See SR 3.6.5.5 and SR 3.6.5.6, above, for further discussion of the basis for the 24 month Frequency.

SR 3.6.5.8

With the spray header drained, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criteria 37, 38, 41, 42, 49, 52, and 58 through 61 issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. USAR Section 6.4.

3. USAR, Section 14.5.

4. USAR, Section 6.3.

5. USAR, Section 5.2.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Spray Additive System

BASES

BACKGROUND

The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a design basis accident (DBA).

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a DBA. It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Because of its stability when exposed to radiation and elevated temperature, sodium hydroxide (NaOH) is the spray additive used at Prairie Island. The NaOH added to the spray also ensures a pH value of between 8.5 and 10.5 in the spray and greater than 7.0 in the solution recirculated from the containment sump (Ref. 1). These pH levels minimize the evolution of iodine as well as the occurrence of chloride and caustic stress corrosion on mechanical systems and components.

The spray additive tank contains at least 2590 gallons of solution with a sodium hydroxide concentration of 9% to 11% by weight.

The Spray Additive System consists of one spray additive tank, two parallel redundant control valves in the line between the additive tank and the containment spray pump suction header, instrumentation, and recirculation pumps. The NaOH solution is added to the spray water by gravity feed at a fixed ratio to the refueling water storage tank (RWST) flow at the suction of the containment spray pumps. Because of the hydrostatic balance between the two tanks, the flow rate of the NaOH is controlled by the volume per foot of height ratio of the two tanks. This ensures a spray mixture pH that is ≥ 8.5 and ≤ 10.5.

BASES

BACKGROUND (continued)	The Containment Spray System actuation signal opens the valves from the spray additive tank to the spray pump suctions. The 9 wt.% to 11 wt.% NaOH solution is drawn into the spray pump suctions. The percent solution and volume of solution sprayed into containment ensures a long term containment sump pH of ≥ 7.0 and ≤ 10.5. This ensures the continued iodine retention effectiveness of the sump water during the recirculation phase and also minimizes the occurrence of chloride induced stress corrosion cracking of the stainless steel recirculation piping.
APPLICABLE SAFETY ANALYSES	The Spray Additive System is essential to the removal of airborne iodine within containment following a DBA. Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its licensing basis value volume for the first 24 hours following the accident. The DBA response time assumed for the Spray Additive System is the same as for the Containment Spray System and is discussed in the Bases for LCO 3.6.5, "Containment Spray and Cooling Systems." The DBA analyses assume that one train of the Containment Spray System/Spray Additive System is inoperable and that the active spray additive tank volume is added to the remaining Containment Spray System flow path. The Spray Additive System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The Spray Additive System is necessary to reduce the release of radioactive material to the environment in the event of a DBA. This system provides NaOH which mixes into the spray flow until the end of the injection phase to raise the average spray solution pH to a level conducive to iodine removal, namely, to between 8.5 and 10.5.

BASES

LCO (continued)	This pH range maximizes the effectiveness of the iodine removal mechanism without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components. The Spray Additive System is considered OPERABLE when: a. The volume of the spray additive solution is $\geq$ 2590 gal and the concentration is $\geq$ 9 weight percent and $\leq$ 11 weight percent; b. Two flow paths from the spray additive tank to the containment spray pump suction header are OPERABLE; c. Manual valves are properly positioned and automatic valves are capable of activating to their correct positions; and d. Piping, valves, instrumentation, and controls for the required flow paths are OPERABLE.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the iodine fission product inventory prior to release to the environment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODE 5 or 6.
ACTIONS	<u>A.1</u> If the Spray Additive System is inoperable, it must be restored to OPERABLE within 24 hours. The pH adjustment of the

BASES

ACTIONS

A.1 (continued)

Containment Spray System flow for corrosion protection and iodine removal enhancement is reduced in this condition. The Containment Spray System would still be available and would remove some iodine from the containment atmosphere in the event of a DBA. The 24 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst case DBA occurring during this period.

B.1 and B.2

If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows 48 hours for restoration of the Spray Additive System in MODE 3 and 36 hours to reach MODE 5. This is reasonable when considering the reduced driving force in MODE 3 for the release of radioactive material from the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment of Spray Additive System manual, power operated, and automatic valves in the spray additive flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1 (continued)

prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.6.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is indicated and alarmed in the control room, so that there is high confidence that a substantial change in level would be detected.

SR 3.6.6.3

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown that these components usually pass the Surveillance when performed. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 6.4.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Hydrogen Recombiners

BASES

BACKGROUND

The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen oxygen reaction.

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors," hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammable limits would not be reached until several days after a design basis accident (DBA).

Two 100% capacity independent hydrogen recombiner systems are provided. Each consists of controls located in the auxiliary building, a power supply and a recombiner. Recombination is accomplished by heating a hydrogen air mixture above 1150° F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.0 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Features bus, and is provided with a separate power panel and control panel.

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.0 v/o following a DBA. This control would prevent a containment wide hydrogen burn, thus ensuring the pressure and temperature assumed in the analyses are

BASES

APPLICABLE SAFETY ANALYSES (continued)

not exceeded. The limiting DBA relative to hydrogen generation is a LOCA.

Hydrogen may accumulate in containment following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 1 are used to maximize the amount of hydrogen calculated.

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 3.5 v/o about 10 days after the LOCA and 4.0 v/o about 6 days later if no recombiner was functioning (Ref. 2). Initiating the hydrogen recombiners when the primary containment hydrogen concentration reaches 3.5 v/o will maintain the hydrogen concentration in the primary containment below flammability limits.

The hydrogen recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed

BASES

APPLICABLE SAFETY ANALYSES (continued)	above, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.0 v/o (Ref. 2). The hydrogen recombiners satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).
LCO	Two hydrogen recombiners must be OPERABLE. This ensures operation of at least one hydrogen recombiner in the event of a worst case single active failure. A hydrogen recombiner is considered OPERABLE when its heater, power supply and controls, are OPERABLE. Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.
APPLICABILITY	In MODES 1 and 2, two hydrogen recombiners are required to control the hydrogen concentration within containment below its flammability limit of 4.0 v/o following a LOCA, assuming a worst case single active failure. In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4. In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations in these MODES. Therefore, hydrogen recombiners are not required in these MODES.

BASES (continued)

ACTIONS

A.1

With one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

Required Action A.1 has been modified by a Note that states the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

B.1

If the inoperable hydrogen recombiner cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Performance of a system functional test for each hydrogen recombiner ensures the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR verifies that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes. After reaching 700°F, the power is increased to maximum power for approximately 2 minutes and power is verified to be ≥ 60 kW.

Operating experience has shown that these components usually pass the Surveillance when performed. Therefore, the 24 month Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.7.2

This SR ensures there are no physical problems that could affect recombiner operation (such as loose wiring or structural connections, or deposits of foreign materials). Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failure involves loss of power, blockage of the internal flow, missile impact, etc.

A visual inspection is sufficient to determine abnormal conditions that could cause such failures. The 24 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

SR 3.6.7.3

This SR requires performance of a resistance to ground test for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms.

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.6.7.3</u> (continued) The 24 month Frequency for this Surveillance was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.
REFERENCES	1. Regulatory Guide 1.7, dated 3/10/71. 2. USAR, Section 5.4.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Vacuum Breaker System

BASES

BACKGROUND

The purpose of the Vacuum Breaker System is to protect the containment vessel against negative pressure (i.e., a lower pressure inside than outside). Excessive negative pressure inside containment can occur if there is an inadvertent actuation of containment cooling features, such as the Containment Spray System or Containment Cooling System. Multiple equipment failures or human errors are necessary to cause inadvertent actuation of these systems.

The containment pressure vessel contains two 100% vacuum breaker trains that protect the containment from excessive external loading.

The characteristics of the vacuum breakers and their locations in the containment pressure vessel are as follows:

Two vacuum breakers are used in each of two large vent lines which permit air to flow from the shield building annulus into the reactor containment vessel. The vacuum breakers consist of an air to close, spring loaded to open butterfly valve and a self-actuated horizontally installed, swinging disc check valve. An air accumulator is provided for each of the air-operated vacuum breakers to allow vacuum breaker operation in the event of a loss of instrument air. The vent lines enter the containment vessel through independent and widely separated containment penetration nozzles. The vacuum breakers serve dual functions in that they are also required to isolate containment following an accident if containment becomes pressurized greater than negative 0.2 psid relative to the shield building annulus.

BASES (continued)

APPLICABLE SAFETY ANALYSES

Design of the Vacuum Breaker System involves calculating the effect of inadvertent actuation of containment cooling features, which can reduce the atmospheric temperature (and hence pressure) inside containment (Ref. 1). Conservative assumptions are used for all the relevant parameters in the calculation: for example, for the Containment Spray System, the minimum spray water temperature, maximum initial containment temperature, maximum spray flow, all spray trains operating, all four containment fan units operating with maximum cooling water flow rate with minimum inlet water temperature, etc. The resulting containment pressure versus time is calculated, including the effect of the opening of the vacuum relief lines when their negative pressure setpoint is reached. It is also assumed that one valve fails to open.

The containment shell was designed for an external pressure load equivalent to 0.8 psi greater than the internal pressure. The inadvertent actuation of the containment cooling features was analyzed to determine the resulting reduction in containment pressure. The analysis shows that one vacuum breaker train will terminate this transient before 0.8 psi pressure differential is reached.

The Vacuum Breaker System must also perform the containment isolation function in a containment high pressure event. For this reason, the system is designed to take the full containment positive design pressure and the environmental conditions (temperature, pressure, humidity, radiation, chemical attack, etc.) associated with the containment DBA.

The vacuum relief valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

The LCO establishes the minimum equipment required to accomplish the vacuum relief function following the inadvertent actuation of containment cooling features. Two 100% vacuum breaker trains are required to be OPERABLE to ensure that at least one is available, assuming one or both valves in the other line fail to open.

A vacuum breaker train is OPERABLE when both valves, including air supplies, instrumentation, controls, and actuating and power circuits, are OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the containment cooling features, such as the Containment Spray System, are required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside containment could occur whenever these systems are required to be OPERABLE due to inadvertent actuation of these systems. Therefore, the vacuum breaker trains are required to be OPERABLE in MODES 1, 2, 3, and 4 to mitigate the effects of inadvertent actuation of the Containment Spray System, or Containment Cooling System.

In MODES 5 and 6, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations of these MODES. The Containment Spray System, and Containment Cooling System are not required to be OPERABLE in MODES 5 and 6. Therefore, maintaining OPERABLE vacuum relief valves is not required in MODE 5 or 6.

BASES (continued)

ACTIONS

A.1 and A.2

When the vacuum relief function of one vacuum breaker train is inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The allowed Completion Time is reasonable considering the redundancy of the other vacuum breaker train, its reliable vacuum relief capability due to the passive design and the low probability of an event requiring use of the Vacuum Breaker System during this time.

B.1 and B.2

If the vacuum breaker train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply.

To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

This SR requires verification that each automatic function of each vacuum breaker train actuates as required to perform its safety function. Testing shall include demonstration that an actual or simulated containment vacuum equal to or less than 0.5 psi will open the air-operated valve and an actual or simulated containment isolation signal with containment pressure greater than negative 0.2 psid relative to the shield building annulus will close the valve. The 92 day Frequency is based on engineering judgment and has been shown to be acceptable through operating experience.

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.6.8.2 This SR requires the performance of a CHANNEL CALIBRATION. A CHANNEL CALIBRATION is performed every 24 months, or approximately at every refueling. Operating experience has shown that these components usually pass the Surveillance when performed.
REFERENCES	1. USAR, Section 5.2.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 Shield Building Ventilation System (SBVS)

BASES

BACKGROUND

As described in the USAR the SBVS is required by AEC GDC 70, "Control of Releases of Radioactivity to the Environment" (Ref. 1), to ensure that radioactive materials that leak from the primary containment into the shield building (secondary containment) following a design basis accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The containment has a secondary containment called the shield building, which is a concrete structure that surrounds the steel primary containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects a portion of the containment leakage following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The SBVS establishes a negative pressure in the annulus between the shield building and the steel containment vessel following a DBA. Filters in the system then control the release of radioactive contaminants to the environment. Shield building OPERABILITY is required to ensure retention of primary containment leakage and proper operation of the SBVS.

The SBVS consists of two separate and redundant trains. Each train includes a heater, a prefilter, moisture separators, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of radioiodines, a recirculation fan and an exhaust fan. Ductwork, valves and/or dampers, and instrumentation also form part of the system. The ventilation system for each shield building includes a vent stack which penetrates the shield building dome and discharges to the atmosphere. The moisture separators function to reduce the moisture content of the airstream. The HEPA filter and the charcoal adsorber section are credited in the analysis. The

BASES

BACKGROUND (continued)

system initiates and maintains a negative air pressure in the shield building by means of filtered exhaust ventilation of the shield building following receipt of a safety injection (SI) signal. The system is described in Reference 2.

The prefilters remove large particles in the air, and the moisture separators remove entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. Heaters are included to reduce the relative humidity of the airstream. Continuous operation of each train, for at least 10 hours per month, with heaters on, reduces moisture buildup on their HEPA filters and adsorbers.

The SBVS reduces the radioactive content in the shield building atmosphere following a DBA. Loss of the SBVS could cause site boundary doses, in the event of a DBA, to exceed the values given in the licensing basis.

APPLICABLE SAFETY ANALYSES

The SBVS design basis is established by the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. 3) assumes that only one train of the SBVS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from containment is determined for a LOCA.

The modeled SBVS actuation in the safety analyses is based upon a worst case response time following an SI initiated at the limiting setpoint. The total response time, from accident initiation to attaining a negative pressure in the shield building, is less than 4.5 minutes. This response time bounds the signal delay, diesel generator startup and sequencing time, system startup time, and time for the system to attain the required pressure after starting.

The SBVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

In the event of a DBA, one SBVS train is required to provide the minimum particulate iodine removal assumed in the safety analysis. Two trains of the SBVS must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single active failure.

A train of SBVS is OPERABLE when its associated:

a. Recirculation and exhaust fan are OPERABLE;

b. HEPA filter and charcoal adsorber are capable of passing their design flow and performing their filtration function;

c. Manual valves and dampers are properly positioned and automatic valves and dampers are capable of activating to their correct positions; and

d. Heater, ductwork, valves, dampers, instrumentation, and controls for the required flow path are OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could lead to fission product release to containment that leaks to the shield building. The large break LOCA, on which this system's design is based, is a full power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and Reactor Coolant System pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the SBVS is not required to be OPERABLE.

BASES (continued)

ACTIONS

A.1

With one SBVS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. In this degraded condition, the remaining components are capable of providing 100% of the iodine removal needs after a DBA. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SBVS train and the low probability of a DBA occurring during this period. The Completion Time is adequate to make most repairs.

B.1 and B.2

If the SBVS train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

Operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 10 continuous hours eliminates moisture on the adsorbers and HEPA filters. Experience from filter testing indicates that the 10 hour period is adequate for moisture elimination on the adsorbers and HEPA filters. Periodic operation also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls, the two train redundancy available, and the iodine removal capability of the Containment Spray System.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.9.2

This SR verifies that the required SBVS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.9.3

The automatic startup ensures that each SBVS train responds properly. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown that these components usually pass the Surveillance when performed. Therefore the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the SR interval was developed considering that the SBVS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.9.1.

SR 3.6.9.4

The SBVS isolation dampers are tested to verify OPERABILITY. The dampers are in the closed position during normal plant operation and must reposition for accident operation to draw air through the filters. The 24 month Frequency is considered to be acceptable based on damper reliability and design, mild environmental conditions in the vicinity of the dampers, and the fact that operating experience has shown that the dampers usually pass the Surveillance when performed.

BASES (continued)

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criterion 70, issued for comment July 10, 1967, as referenced in USAR Section 1.2.
2. USAR, Section 5.3.
3. USAR, Section 14.9.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.10 Shield Building

BASES

BACKGROUND

The shield building is a concrete structure that surrounds the steel containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects a portion of the containment leakage that may occur following a design basis accident (DBA). This space also allows for periodic inspection of the outer surface of the steel containment vessel. The shield building provides biological shielding for DBA conditions, protects the containment vessel from low temperatures, adverse atmospheric conditions and external missiles, and provides the means for collecting and filtering containment fission product leakage following a DBA (Ref. 1).

Following a DBA the Shield Building Ventilation System (SBVS) establishes a negative pressure in the annulus between the shield building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment. The shield building is required to be OPERABLE to ensure retention of containment leakage and proper operation of the SBVS.

APPLICABLE SAFETY ANALYSES

The design basis for shield building OPERABILITY is a loss of coolant accident (LOCA). Maintaining shield building OPERABILITY ensures that the release of radioactive material from the containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analyses.

The shield building satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

Shield building OPERABILITY must be maintained to ensure proper operation of the SBVS and to limit radioactive leakage

BASES

LCO (continued)	from the containment to those paths and leakage rates assumed in the accident analyses. The shield building is OPERABLE when: a. At least one door in each access opening is closed including when the access opening is being used for normal transit entry and exit; and b. The shield building equipment opening is closed.
APPLICABILITY	Maintaining shield building OPERABILITY prevents leakage of radioactive material from the shield building. Radioactive material may enter the shield building from the containment following a DBA. Therefore, shield building OPERABILITY is required in MODES 1, 2, 3, and 4 when a DBA could release radioactive material to the containment atmosphere. In MODES 5 and 6, the probability and consequences of a DBA are low due to the Reactor Coolant System temperature and pressure limitations in these MODES. Therefore, shield building OPERABILITY is not required in MODE 5 or 6.
ACTIONS	<u>A.1</u> In the event shield building OPERABILITY is not maintained, shield building OPERABILITY must be restored within 24 hours. Twenty-four hours is a reasonable Completion Time considering the limited leakage design of containment and the low probability of a DBA occurring during this time period.

BASES

ACTIONS (continued)

B.1 and B.2

If the shield building cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1

Maintaining shield building OPERABILITY requires verifying one door in the access opening closed. Each access opening into the shield building contains one inner and one outer door. The intent is to not breach the shield building boundary at any time when the shield building boundary is required. This is achieved by maintaining the inner or outer portion of the barrier closed at all times. However, all shield building access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening. The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available to the operator.

SR 3.6.10.2

The SBVS produces a negative pressure to prevent leakage from the building. SR 3.6.10.2 verifies that the shield building can be rapidly drawn down to -2.00 inch water gauge and maintains a pressure equal to or more negative than -1.82 inches of water gauge in the

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.10.2 (continued)

annulus after the recirculation dampers open and equilibium is established. Equilibrium negative pressure equal to or more negative than -1.82 inches water gage is that predicted for non-accident conditions and leakage equal to 75% of the maximum allowable shield building inleakage (Reference 2). Establishment of this pressure is confirmed by SR 3.6.10.2, which demonstrates that the shield building can be drawn down to ≤ -2.0 inches of vacuum water gauge in the annulus using one SBVS train.

The primary purpose of this SR is to ensure shield building integrity. The secondary purpose of this SR is to ensure that the SBVS being tested functions as designed. The inoperability of the SBVS train does not necessarily constitute a failure of this Surveillance relative to the shield building OPERABILITY.

The 31 day Frequency provides assurance that shield building integrity is maintained and the system will function as required.

REFERENCES

1. USAR, Section 5.3.

2. "Report to the United States Nuclear Regulatory Commission Division of Operating Reactors - Prairie Island Containment Systems Special Analyses", dated April 9, 1976.

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the USAR (Ref. 1). The MSSVs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-1 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip.

Normal functioning of a MSSV is expected to involve some "simmering" which does not make the valve inoperable.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure for any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis. The accident analysis requires five MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 102% RTP.

BASES

APPLICABLE SAFETY ANALYSES (continued)	By relieving steam, the MSSVs prevent RCS overpressurization. The limiting events, described in the USAR (Ref. 3), that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, such as the full power turbine trip without steam dump, and increasing core heat flux events, such as the rod cluster control assembly (RCCA) withdrawal at power. The safety analyses demonstrate that the transient response for turbine trip occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. If a minimum (least negative or most positive) reactivity feedback is assumed, the reactor is tripped on high pressurizer pressure. In this case, the pressurizer safety valves open, and RCS pressure remains below 110% of the design value. The MSSVs also open to limit the secondary steam pressure. The transient response for the slow and fast RCCA withdrawal at power events also present no hazard to the integrity of the RCS or the Main Steam System. Diverse reactor trip inputs from nuclear instrumentation and pressurizer level and pressure are assumed to shut down the reactor when the associated trip setpoint is reached. In this analysis, the pressurizer safety valves open and RCS pressure remains below 110% of the design value. The MSSVs also open to limit the secondary steam pressure. The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2. The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances, relieve steam generator overpressure, and close when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

BASES

LCO (continued)

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

In MODES 4, 5, and 6, there are no credible transients requiring the MSSVs.

The energy content in the steam generators is sufficiently low in MODES 5 and 6 that they cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

A.1

With one MSSV inoperable, restore OPERABILITY of the inoperable MSSV within 4 hours. The 4 hours is a reasonable time due to the low probability of an event or transient occurring during this time requiring MSSV operation.

Continued operation with less than all five MSSVs OPERABLE for each steam generator is not permitted since safety analyses supporting such operation have not been performed.

B.1 and B.2

If the MSSV cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within

BASES

ACTIONS

B.1 and B.2 (continued)

12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-1 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to within a nominal ± 1% of their setpoint during the Surveillance. The lift settings, according to Table 3.7.1-1, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. USAR, Section 11.4.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. USAR, Section 14.4.
4. ASME, Boiler and Pressure Vessel Code, Section XI.
5. ANSI/ASME OM-1-1987.

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a main steam line break (MSLB). MSIV closure terminates flow from the unaffected (intact) steam generators.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Steam Dump System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal generated by any of the following signals:

a. High-High Containment Pressure;

b. High Steam Flow and Low-Low T_{avg} with Safety Injection; and

c. High-High Steam Flow with Safety Injection.

The MSIVs fail closed on loss of air.

Each MSIV has a MSIV bypass valve. These valves are normally used for warming steam lines and equalizing pressure across the MSIVs. These bypass valves are normally closed at power.

The MSIVs and MSIV bypass valves may be operated manually.

BASES

BACKGROUND (continued)	In addition to the fast-closing stop valve, each steam line has a downstream non-return check valve (NRCV). The four valves (one MSIV and one NRCV in each of two lines) prevent blowdown of more than one steam generator for any break location even if one valve fails to close. A description of the MSIVs and NRCVs is found in the USAR (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the USAR (Ref. 2). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV or NRCV to close).

The limiting case for the containment analysis is the main steam line break (MSLB) inside containment, with offsite power available following turbine trip, and failure of a safeguards train. At lower power, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.

The analysis of several different SLB events are performed to demonstrate that the acceptance criteria listed in the USAR are satisfied.

Events evaluated include:

a. Containment response due to a large SLB inside of containment;

b. Core response due to a large SLB inside of containment;

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Small SLB; and

d. Core response due to a SLB outside of containment to support the voltage-based steam generator tube repair criteria (Ref. 3).

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

a. A MSLB inside containment. For this accident scenario, steam is discharged into containment from both steam generators until the NRCV on the broken line (or MSIV on the intact line) closes. After the valve closes, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header between the closed valve and the affected steam generator. Closure of the NRCV in the affected line (or the MSIV in the intact line) isolates the break from the unaffected steam generator.

b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the NRCV in the affected line (or the MSIV in the intact line) isolates the break and limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.

d. Following a steam generator tube rupture, closure of the MSIV downstream of the ruptured steam generator isolates the ruptured steam generator from the intact steam generator to minimize radiological releases.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

This LCO requires that both MSIVs be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on a main steam isolation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, normally the MSIVs are closed, and the steam generator energy is low. In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the MSIV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs and considering the redundancy of the NRCV.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional passive means for containment isolation.

BASES

ACTIONS (continued)

B.1

If the MSIV cannot be restored to OPERABLE status within 8 hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition C would be entered unless both MSIVs are closed. The Completion Times are reasonable, based on operating experience, to reach MODE 2 in an orderly manner without challenging unit systems.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIV may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The 8 hour Completion Time is consistent with that allowed in Condition A for one MSIV inoperable.

For an inoperable MSIV that cannot be restored to OPERABLE status within the specified Completion Time, but is closed, the inoperable MSIV must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

BASES

ACTIONS (continued)

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 5 seconds. The MSIV isolation time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of valve closure when the unit is generating power. As the MSIVs are not tested at power, they are deferred from the ASME Code (Ref. 5) requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program.

This test is conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.2

This SR verifies each MSIV can close on an actual or simulated main steam isolation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage.

The Frequency of MSIV testing is every 24 months. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 11.7.
2. USAR, Section 14.5.
3. License Amendment 133/125, issued November 18, 1997, "Voltage-based Steam Generator Tube Repair Criteria."
4. 10 CFR 100.11.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Regulation Valves (MFRVs) and MFRV Bypass Valves

BASES

BACKGROUND

The MFRVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a steam or feedwater line break. The key safety-significant functions of the MFRVs are to prevent:

a. Overfill of the steam generators;

b. Excessive cooldown of the Reactor Coolant System; and

c. Overpressurization of the containment following a main feedwater line break (FWLB) or main steam line break (MSLB).

Closure of the MFRVs and associated bypass valves terminates flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring downstream of the main feedwater isolation valves (MFIVs) or MFRVs. Credit is taken for only the MFRVs and the MFRV bypass valves since the closure times are shorter than those of the MFIVs. The MFIVs are treated solely as containment isolation valves in accordance with LCO 3.6.3, "Containment Isolation Valves". Check valves in the feedwater lines terminate FWLBs upstream of the MFIVs and MFRVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFRVs will be mitigated by their closure. Closure of the MFRVs and MFRV bypass valves effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs.

The MFRVs, MFRV bypass valves, and piping upstream of the MFIVs are nonsafety related. In the event of a secondary side pipe rupture inside containment, the MFRVs, MFRV bypass valves,

BASES

BACKGROUND (continued)

check valves and the main feedwater pump trip limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops.

One MFRV and its MFRV bypass valve are located on each MFW line, outside but close to containment. The check valves are located upstream of the AFW injection point so that AFW may be supplied to the steam generators following MFIV or MFRV closure. The piping volume from the valves to the steam generators must be accounted for in calculating mass and energy releases. This line must be refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

The MFRVs and MFRV bypass valves close due to the following automatic feedwater isolation (FWI) signals:

a. Low T_{avg} coincident with reactor trip (P-4) (MFRV only);

b. Steam generator water level-high high signal; and

c. Safety injection.

The MFRVs and MFRV bypass valves may also be operated manually.

In addition to the MFIVs, the MFRVs and MFRV bypass valves, a check valve inside containment is available. The check valve isolates the feedwater line, penetrating containment, and ensures that the consequences of events do not exceed the capacity of the containment heat removal systems.

A description of the MFRVs is found in the USAR (Ref. 1).

BASES (continued)

APPLICABLE SAFETY ANALYSES

The design basis of the MFRVs is established by the analyses for the large Main Steam Line Break (MSLB). Closure of the MFRVs and associated bypass valves, and trip of the main feedwater pumps may be relied on to terminate feedwater flow during a MSLB for the core and containment response analyses.

The MSLB core and containment response analyses bound the accident analysis for the large FWLB. Some leakage through the MFRVs and associated bypass valves is anticipated when control board instrumentation indicates that the valves have closed. This leakage has been conservatively bounded by the MSLB analyses.

Failure of a MFRV or the MFRV bypass valves to close following a MSLB can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following a MSLB or FWLB event.

The MFRVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO ensures that the MFRVs and the MFRV bypass valves will isolate MFW flow to the steam generators, following a FWLB or MSLB.

This LCO requires that two MFRVs and associated MFRV bypass valves be OPERABLE. The MFRVs and the associated MFRV bypass valves are considered OPERABLE when feedwater isolation times are within limits and they close on a FWI signal. When control board instrumentation indicates that these valves have fully closed, the valves are OPERABLE since leakage through the closed valves has been conservatively bounded by the MSLB analyses, and therefore, they are performing their safety function.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or

BASES

LCO (continued)	FWLB inside containment. Since a feedwater isolation signal on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.
APPLICABILITY	The MFRVs and the MFRV bypass valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. In MODES 1, 2, and 3, the MFRVs and the MFRV bypass valves are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed, they are already performing their safety function. In MODES 4, 5, and 6, steam generator energy is low. In addition, the MFRVs and the MFRV bypass valves are normally closed since MFW is not required.
ACTIONS	The ACTIONS table is modified by two Notes. Note 1 specifies separate Condition entry is allowed for each valve. Note 2 specifies LCO 3.0.4 does not apply. <u>A.1 and A.2</u> With one MFRV in one or both flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close and place in manual or to isolate flow through inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function. Similarly, if the feedwater flow path to containment is isolated using, as an example, the MFIV, the required safety function is being met.

BASES

ACTIONS

A.1 and A.2 (continued)

The 72 hour Completion Time takes into account the redundancy afforded by the remaining valves in the feedwater line and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFRVs, that are closed and in manual or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls to ensure that the valves are closed or isolated.

B.1 and B.2

With one MFRV bypass valve in one or both flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close and place in manual or to isolate flow through inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function. Similarly, if the feedwater flow path to containment is isolated using, as an example, the MFIV, the required safety function is being met.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining valves in the feedwater line and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

BASES

ACTIONS

B.1 and B.2 (continued)

Inoperable MFRV bypass valves that are closed and placed in manual or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

C.1 and C.2

If the MFRV(s) or the MFRV bypass valve(s) cannot be restored to OPERABLE status, closed, isolated, or the flow path through the valve isolated within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFRV and MFRV bypass valve is within limits set by the Inservice Testing Program. The MFRV isolation times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. This is consistent with the ASME Code (Ref. 2) periodic stroke requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the Inservice Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.3.2

This SR verifies that each MFRV and MFRV bypass valve can close on an actual or simulated FWI signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage.

The Frequency for this SR is every 24 months. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section 11.9.

2. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.7 PLANT SYSTEMS

B 3.7.4 Steam Generator (SG) Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The SG PORVs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Dump System to the condenser be unavailable, as discussed in the USAR (Ref. 1). Cooldown is performed in conjunction with the Auxiliary Feedwater System providing makeup water to the steam generators.

One SG PORV line is provided for each steam generator. Each SG PORV line consists of one SG PORV and an associated block valve.

The upstream manual block valves permit SG PORV testing at power and provide an alternate means of isolation. The SG PORVs are equipped with pneumatic controllers to permit control of the cooldown rate.

A description of the SG PORVs is found in References 1 and 2.

APPLICABLE SAFETY ANALYSES

Automatic operation of the SG PORVs is not credited in the safety analyses. Rather, the SG PORVs may provide mitigation for accidents involving use of main steam safety valves.

In the steam generator tube rupture (SGTR) accident analysis presented in Reference 2, the SG PORV in the unaffected steam generator is assumed to be used by the operator to cool down the unit for accidents accompanied by a loss of offsite power. Prior to operator actions to cool down the unit, the main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a SGTR event, the operator is required to perform a limited cooldown to establish adequate subcooling as a necessary

BASES

APPLICABLE SAFETY ANALYSES (continued)	step prior to terminating the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for a SGTR is more critical than the time required to cool down for this event and also for other accidents. The SG PORVs are equipped with manual block valves in the event a SG PORV spuriously fails open or fails to close during use. The SG PORVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Two SG PORV lines are required to be OPERABLE to ensure that at least one SG PORV is available to conduct a unit cooldown following a SGTR. Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an event in which the condenser is unavailable for use with the Steam Dump System. A SG PORV is considered OPERABLE when it is capable of being remotely operated and when its associated block valve is open.
APPLICABILITY	In MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal, the SG PORVs are required to be OPERABLE. In MODE 5 or 6, a SGTR is not a credible event.

BASES (continued)

ACTIONS

A.1

With one required SG PORV line inoperable, action must be taken to restore OPERABLE status within 7 days.

The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE SG PORV lines, Steam Dump System, and MSSVs.

Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply.

B.1

With two SG PORV lines inoperable, action must be taken to restore one SG PORV to OPERABLE status. Since the block valve can be closed to isolate a SG PORV, some repairs may be possible with the unit at power.

The 1 hour Completion Time allows time to plan an orderly shutdown of the unit and is reasonable, based on the availability of the Steam Dump System and MSSVs, and the low probability of an event occurring during this period that would require the SG PORV lines.

C.1 and C.2

If the SG PORV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon steam generator for heat removal, within 12 hours.

BASES

ACTIONS

C.1 and C.2 (continued)

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR ensures that the SG PORVs are tested through a full control cycle in accordance with the Inservice Testing Program. The SG PORV is isolated by the block valve for this test. Performance of inservice testing or use of a SG PORV during a unit cooldown may satisfy this requirement.

Operating experience has shown that these components usually pass the Surveillance when performed in accordance with the Inservice Testing Program. The Frequency is acceptable from a reliability standpoint.

SR 3.7.4.2

The function of the block valve is to isolate a failed open SG PORV. Manually cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement.

Operating experience has shown that these components usually pass the Surveillance when performed. The Frequency is acceptable from a reliability standpoint.

BASES (continued)

REFERENCES

1. USAR, Section 11.4.
2. USAR, Section 14.

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply.

The AFW system is configured into two redundant trains. One train has a turbine driven AFW pump; the other has a motor driven AFW pump. Each AFW pump feeds the designated unit's two steam generators. In addition, each motor driven pump has the capability to be realigned locally to feed the other unit's steam generators.

The AFW pumps take suction from:

a. The nonsafety-related condensate storage tank (CST) supply header (LCO 3.7.6); or

b. The safety-related Cooling Water System (LCO 3.7.8).

The AFW pumps supply water to the steam generator secondary side via connections to the main feedwater (MFW) piping adjacent to the steam generators inside containment.

The steam generators function as a heat sink for core decay heat. The heat load may be dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1), steam generator power operated relief valves (SG PORVs) (LCO 3.7.4), or steam dump valve.

If the main condenser is available, steam may be released via the steam dump valve. Each unit's AFW System consists of:

BASES

BACKGROUND (continued)

a. One motor driven AFW pump;

b. One turbine driven AFW pump;

c. Steam generator AFW motor-operated supply valves; and

d. Steam generator AFW motor-operated throttle valves.

These components are configured to provide a flow path from each pump to both steam generators for the specific unit.

Each motor driven or turbine driven AFW pump can provide 100% of the required AFW flow capacity to the steam generators, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system.

The turbine driven AFW pump receives steam from both main steam lines upstream of the main steam isolation valves. Each steam feed line will supply 100% of the requirements of the turbine driven AFW pump. An air operated valve downstream of the motor operated valves from each loop allows passage of steam to the turbine driven AFW pump when required. The air supply to the valve is controlled by a normally open DC solenoid valve designed such that failure of either the air supply or control power would cause the respective valve to open, starting the turbine driven AFW pump.

The AFW System is capable of supplying feedwater to the steam generators during normal unit operation in MODES 2 and 3. One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions.

The AFW System is designed to supply sufficient water to the steam generators to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies

BASES

BACKGROUND (continued)	sufficient water to cool the unit to RHR entry conditions, with steam released through the SG PORVs or steam dump valve. The following safety signals automatically initiate an AFW pump start signal: a. Low-low water level in either steam generator; and b. Safety injection. Additionally, the following signals initiate an AFW pump start signal: a. Trip of both main feedwater pumps (bypassed during startup and shutdown operation); b. Loss of both 4 kV normal buses (turbine driven AFW pump only); and c. Manually either local or remote. Depending on pump type, the motor will start or the turbine steam admission air operated control valve will open. The AFW System is discussed in the USAR (Ref. 1).
APPLICABLE SAFETY ANALYSES	The AFW System mitigates the consequences of any event involving loss of normal feedwater. The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus margin for uncertainty and accumulation.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The limiting plant condition which imposes safety-related performance requirements on the design of the AFW System is the loss of MFW as described in References 1 and 3. The low-low steam generator level signal automatically actuates the motor and turbine driven AFW pumps and associated air operated valve and controls when required to ensure an adequate feedwater supply to the steam generators during loss of offsite power. Normally open motor operated valves are provided for each AFW line to allow throttling of the AFW flow from each AFW pump to each steam generator when required. The AFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Two independent AFW pumps in two diverse trains are required to be OPERABLE to ensure the availability of decay heat removal capability for all events accompanied by a loss of main feedwater and a single failure. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that one motor driven AFW pump be OPERABLE and capable of supplying AFW to both steam generators. The turbine driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to both steam generators. The piping, valves, instrumentation, and controls in the required flow paths, required for

BASES

LCO (continued)	the system to perform the safety related function, also are required to be OPERABLE. The normal (Condensate Storage Tanks (CSTs)) and backup (Cooling Water System) water supplies to the AFW pumps must also be OPERABLE. OPERABILITY requirements for the CSTs are specified in LCO 3.7.6, "Condensate Storage Tanks (CSTs)." The LCO is modified by two Notes. The first Note indicating that an AFW train may be considered OPERABLE during alignment and operation for steam generator level control if capable of being manually realigned to the AFW mode of operation. The second Note indicating that an AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. This is because of the reduced heat removal requirements and short period of time in MODE 4 during which the AFW is required and the insufficient steam available in MODE 4 to power the turbine driven AFW pump. During operation in MODES 2 and 3, the AFW pump discharge motor operated valves used for throttling may be less than full open. The Shutdown-Auto mode of control may be used during such operations. This control mode bypasses the AFW pump start due to both MFW pumps being tripped or shutdown.
APPLICABILITY	In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to provide heat removal. In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory, lost as the unit cools to MODE 4 conditions. In MODE 4 the AFW System may be used for heat removal via the steam generators. In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required to perform a safety function.

BASES (continued)

ACTIONS

A.1

If one of the two steam supplies to the turbine driven AFW train is inoperable, or if a turbine driven pump is inoperable while in MODE 3 immediately following refueling, action must be taken to restore the inoperable equipment to an OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. For the inoperability of a steam supply to the turbine driven AFW pump, the 7 day Completion Time is reasonable since there is a redundant steam supply line for the turbine driven pump;

b. For the inoperability of a turbine driven AFW pump while in MODE 3 immediately subsequent to a refueling outage, the 7 day Completion Time is reasonable due to the minimal decay heat levels in this situation; and

c. For both the inoperability of a steam supply line to the turbine driven pump and an inoperable turbine driven AFW pump while in MODE 3 immediately following a refueling outage, the 7 day Completion Time is reasonable due to the availability of the redundant OPERABLE motor driven AFW pump, and due to the low probability of an event requiring the use of the turbine driven AFW pump.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector

BASES

ACTIONS

A.1 (continued)

between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

Condition A is modified by a Note which limits the applicability of the Condition when the unit has not entered MODE 2 following a refueling. Condition A allows one AFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.

B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

BASES

ACTIONS (continued)

C.1 and C.2

When Required Action A.1 or B.1 cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODE 1, 2, or 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

D.1

If both AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

BASES

ACTIONS (continued)

E.1

In MODE 4, either the reactor coolant pumps or the RHR Loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops-MODE 4." With one required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR verifies the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths thereby providing assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves.

This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1 (continued)

train(s) inoperable. Since AFW may be used during MODES 2, 3, and 4 operations for steam generator level control, and these manual operations are an accepted function of the AFW system, OPERABILITY (i.e., the intended safety function) continues to be maintained.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Differential pressure is a normal test of centrifugal pump performance required by Section XI of the ASME Code (Ref. 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code, Section XI (Ref. 2) satisfies this requirement. The Inservice Testing Program specifies the Frequency for testing each pump. This test is considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. This deferral is based on the inservice testing requirements not met; all other requirements for OPERABILITY must be satisfied.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated safety injection signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This test is considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

This SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during MODES 2, 3, and 4 operations for steam generator level control, and these manual operations are an accepted function of the AFW system, OPERABILITY (i.e., the intended safety function) continues to be maintained.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4

This SR verifies that the AFW pumps will start when required by demonstrating that each AFW pump starts automatically on an actual or simulated AFW pump start signal. Since this test is performed during unit shutdown, the turbine driven AFW pump is not actually started, but the components necessary to assure it starts on an actual or simulated AFW pump start signal are demonstrated to be OPERABLE. This test is considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

This SR is modified by two Notes. Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. Note 2 states that one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during MODES 2, 3, and 4 operations for steam generator level control, and these manual operations are an accepted function of the AFW system, OPERABILITY (i.e., the intended safety function) continues to be maintained.

BASES (continued)

REFERENCES

1. USAR, Section 11.9.
2. ASME, Boiler and Pressure Vessel Code, Section XI.
3. USAR, Section 14.4.

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tanks (CSTs)

BASES

BACKGROUND

Three 150,000 gallon CSTs (total) are shared via a common header between the 2 units. Unit 1 has 1 tank (11) and Unit 2 has 2 tanks (21 and 22). The CSTs provide a nonsafety grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS).

A backup safety grade source of water is provided by the safety-related portion of the Cooling Water (CL) System (LCO 3.7.8) via either the Emergency Cooling Water Line or the emergency bay sluice gates.

Since water supplied from the CL System is of lower purity, its use is considered less desirable under normal conditions than the higher purity condensate water from the CSTs. However, if needed, the operator can lineup the Cooling Water supply by opening the associated CL supply motor operated valve (MOV) and closing the associated CST supply MOV for each auxiliary feedwater pump.

The CSTs provide a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves, the steam generator power operated relief valves or the atmospheric dump valve. Each AFW pump operates with a continuous recirculation to a CST.

When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the steam dump valve. The condensed steam may be returned to the CSTs by the condensate pump. This has the advantage of conserving condensate while minimizing releases to the environment.

BASES

BACKGROUND (continued)

Although the CSTs are a principal secondary side water source for removing residual heat from the RCS, they are not designed to withstand earthquakes and other natural phenomena, including missiles that might be generated by natural phenomena. However, the backup CL safety-related source is designed to withstand such phenomena.

A description of the CSTs is found in the USAR (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CSTs may provide high purity cooling water to remove decay heat and to cool down the unit following events in the accident analysis as discussed in the USAR (Ref. 2).

The 100,000 gallon CSTs useable volume requirement for each unit in MODE 1, 2, or 3 is sufficient to:

a. Remove the decay heat generated by one reactor in the first 12 hours after shutdown; and

b. Ensure sufficient water is available to cool down a reactor from 547°F to 350°F using natural circulation at 25°F/hour; or

c. Ensure sufficient water is available to hold the unit in MODE 3 for 2 hours, folllowed by a cooldown to RHR entry conditions within the next 6 hours.

These calculations take into account the decay heat and reactor coolant system stored energy (Ref. 1).

The CST satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

The CSTs are considered OPERABLE when the CSTs' contents have at least 100,000 useable gallons per operating unit (MODES 1, 2, or 3).

This basis is established in Reference 2 and exceeds the volume required by the accident analysis.

The OPERABILITY of the CSTs is determined by maintaining the tank level at or above the minimum required level.

APPLICABILITY

In MODES 1, 2, and 3, and MODE 4, when steam generator is being relied upon for heat removal, the CSTs are required to be OPERABLE.

In MODE 5 or 6, the CSTs are not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the CSTs are not OPERABLE (e.g., level is not within limits), the OPERABILITY of the backup safety-related portion of the CL supply should be verified by administrative means within 4 hours and once every 12 hours thereafter. OPERABILITY of the backup safety-related portion of the CL supply must include verification that the flow paths from the backup water supply to the AFW pumps are OPERABLE in accordance with LCO 3.7.8. The CSTs must be restored to OPERABLE status within 7 days.

The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup safety-related portion of the Cooling Water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7 day Completion Time is reasonable, based on an OPERABLE backup

BASES

ACTIONS

A.1 and A.2 (continued)

safety-related portion of the CL supply being available, and the low probability of an event occurring during this time period requiring the CSTs.

B.1 and B.2

If the CSTs cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply.

To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on the steam generator for heat removal, within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CSTs contain the required useable volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST inventory between checks.

Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator of abnormal deviations in the CST level.

BASES (continued)

REFERENCES

1. USAR, Section 11.9.
2. USAR, Section 14.4.

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CC) System

BASES

BACKGROUND

The CC System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CC System also provides this function for various nonessential components, as well as the spent fuel storage pool. The CC System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Cooling Water System, and thus to the environment.

Each unit's CC System is arranged as two parallel cooling loops, and has isolable nonsafety related components. Each safety related train includes a full capacity pump, supply from a common unit-specific surge tank, heat exchanger, piping, valves, and instrumentation.

The CC systems have the capability to be cross-connected between loops and between the two units at the CC pump suction and discharge. This design feature is not used during normal operation but does allow added flexibility of pump and heat exchanger combinations during abnormal conditions.

During operation the outlet CC temperature from the CC heat exchanger is normally maintained between 80°F and 105°F. During operation CC water is circulated through the shell side of the CC heat exchanger and then to the various system components at a maximum temperature of 120°F for 2 hours (Ref. 1).

Each safety related train is powered from a separate bus. A surge tank in the system is provided to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal, and all nonessential

BASES

BACKROUND (continued)

components are isolated. In addition, an automatic low pressure pump start can avert a reactor coolant pump seal failure during a loss of offsite power event (Ref. 1).

Additional information on the design and operation of the system, along with a list of the components served, is presented in the USAR (Ref. 1).

The principal safety related function of the CC System is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System during post accident containment sump recirculation.

APPLICABLE SAFETY ANALYSES

The design basis of the CC System is for one CC train to remove the post loss of coolant accident (LOCA) heat load from the containment sump during the recirculation phase.

During an accident, a design CC inlet temperature to the major heat exchangers of 95°F was assumed. The RHR heat exchanger during DBA long-term conditions is the primary heat load. The required post accident heat removal rate is in the same range as the required rate during MODES 1 or 2, but less than that needed during normal MODE 4 condition. In a normal MODE 4 cooldown from 350°F to 200°F, more equipment is expected to be operating than during a post accident condition or cooldown. The time required to cool from 350°F to 200°F is a function of the number of CC and RHR trains operating (Ref. 1).

The CC System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power. One CC train is sufficient to remove decay heat during subsequent operations.

The CC System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

The CC trains are independent of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a DBA, one CC train is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two trains of CC must be OPERABLE. At least one CC train will operate assuming the worst case single active failure occurs coincident with a loss of offsite power.

A CC train is considered OPERABLE when:

a. The pump and associated surge tank are OPERABLE; and

b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

The isolation of CC from other components or systems may render those components or systems inoperable but does not affect the OPERABILITY of the CC System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CC System is a normally operating system, which must be prepared to perform its post accident safety functions, including but not limited to RCS heat removal, which is achieved by cooling the RHR heat exchanger.

In MODE 5 or 6, the OPERABILITY requirements of the CC System are determined by the systems it supports.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.6,

BASES

ACTIONS

A.1 (continued)

"RCS Loops-MODE 4," be entered if an inoperable CC train results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

If one CC train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CC train is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the CC train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CC flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CC System.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1 (continued)

This SR verifies the correct alignment for manual, air operated, and automatic valves in the CC flow path. This provides assurance that the proper flow paths exist for CC operation.

This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves.

This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. Control room indication may be used to fulfill this SR.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the CC valves on an actual or simulated safety injection actuation signal.

The CC System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This test is considered satisfactory if control board indication and visual observation of the equipment demonstrate that all components have operated properly.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.2 (continued)

The 24 month Frequency is based on engineering judgement and to allow performance of this Surveillance under the conditions that apply during a unit outage. Although this SR may be performed during normal power operation, there may be plant conditions when it is advantageous to perform this Surveillance during a unit outage.

Operating experience has shown that these components usually pass the Surveillance when performed. Therefore, the Frequency is acceptable from a reliability standpoint.

This SR is modified by a Note stating that the SR applies to those valves required to align CC System to support the safety injection or recirculation phases of emergency core cooling.

SR 3.7.7.3

The CC pumps may be actuated by either a safety injection signal or system low pressure. This SR verifies proper automatic operation of the CC pumps on an actual or simulated safety injection actuation signal and verifies proper automatic operation of the CC pumps on an actual or simulated low pressure actuation signal. This test is considered satisfactory if control board indication and visual observation of the equipment demonstrate that all components have operated properly.

The CC System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation.

The 24 month Frequency is based on engineering judgement and to allow performance of this Surveillance under the conditions that apply during a unit outage. Although this SR may be performed during normal power operation, there may be plant conditions when it is advantageous to perform this Surveillance during a unit outage.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.7.3 (continued) Operating experience has shown that these components usually pass the Surveillance when performed. Therefore, the Frequency is acceptable from a reliability standpoint.
REFERENCES	1. USAR, Section 10.4.